NON-PUBLIC?: N
ACCESSION #: 9006140199
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK- UNIT 1 PAGE: 1 OF 11

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP DUE TO LOSS OF FEEDWATER PUMP SPEED
CONTROLLERS
DURING MAINTENANCE
EVENT DATE: 05/09/90 LER #: 90-013-00 REPORT DATE: 06/08/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 048

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At approximately 1504 CDT on May 9, 1990, while using a procedure, written for Mode 5 Cold Shutdown, or Mode 6 Refueling, to calibrate a feedwater pump discharge pressure transmitter, jumpers were installed across the feedwater pump speed controllers while they were being used to maintain feedwater pump speed during Mode 1 Power Operation. Installation of the jumpers caused a coastdown of the feedwater pumps, resulting in a loss of feedwater flow and reduction of steam generator water levels. An automatic reactor thp occurred when the Steam Generator Water Level Low-Low Trip Setpoint was reached. Plant recovery was completed without further incident or unexpected findings.

The root cause of the event has been determined to be an inadequate

review and approval process for certain procedure changes. As corrective actions for this event, a procedure revision will ensure technical reviews and operational impact assessments are performed for mode applicability interpretations. Generic corrective actions will increase sensitivity to the impact that non-safety components can have on an operating plant. The Single Point Failure Analysis, which identifies these components, will be utilized programmatically.

END OF ABSTRACT

1. DESCRIPTION OF WHAT OCCURRED

A. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On May 9, 1990 at 1500, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, operating at 48 percent power.

B. REPORTABLE EVENT DESCRIPTION (INCLUDING DATES AND APPROXIMATE
TIMES OF MAJOR OCCURRENCES)

Event Classification: An event or condition that resulted in an automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

On the morning of May 9, 1990, the feedwater pump discharge pressure transmitter (EIIS:(PT)(SJ)) was identified as indicating about 70 pounds per square inch-gage (PSIG) less than actual feedwater header pressure. A corrective maintenance work order was written to troubleshoot the problem and, if necessary, to calibrate the transmitter. This work order was assigned priority 12, meaning that it would be worked for 24 hours a day to achieve the easiest possible completion. It was decided that the calibration procedure for the transmitter which was often to be performed during Mode 5, Cold Shutdown, or Mode 6, Refueling, would be used as part of the work order To allow the procedure to be performed during Mode 1, a review of technical specifications and commitments was performed at approximately 1300 by an Instrumentation and Control (I&C) support engineer (utility, non-licensed). The review was performed in accordance with the I&C Work Control procedure and determined that there were no technical specification requirements or commitments applicable to the feedwater pump discharge pressure channel. This review was concurred with by the on duty Unit Supervisor (utility, licensed).

After receiving approval from the Unit Supervisor and notifying the Reactor Operator (utility, licensed), the I&C technicians (utility, non-licensed) began to set up for calibrating the transmitter. The second step of the calibration setup instructions was to install jumpers across the output of the Westinghouse feedwater pump speed automatic and manual controllers (EIIS:(SCO)(SJ)). This step was performed at approximately 1504. Because both feedwater pumps (EIIS:(P)(Sj)) were being controlled by the Westinghouse speed control signals, performance of this step caused both feedwater pumps to coast down with a zero speed demand. The coastdown of the feedwater pumps caused a loss of feedwater flow which was annunciated in the control room by several alarms.

The control room reactor operators (utility, licensed), alerted by main control board alarms (EIIS:(ALM)(IB)), attempted to restore feedwater pump speed with the Westinghouse speed controllers in manual. Manual control, however, had no effect on the feedwater pumps with the output of the Westinghouse speed controllers jumpered. Immediately thereafter, the feedwater isolation valves (EIIS:(ISV)(Sj)) closed on anti-water hammer interlock due to low feedwater flow. Approximately 30 seconds later, a reactor operator (utility, licensed) manually started the Train A and Train B motor driven auxiliary feedwater pumps (EIIS:(P)(BA)). With the feedwater isolation valves closed and steam generator water levels already low, the Unit Supervisor determined that recovery was not possible and ordered a manual turbine trip and reactor thp. The turbine was tripped at approximately 1506. The turbine thp caused a shrink in steam generator water level below the steam generator low-low-level reactor trip setpoint, and the reactor tripped automatically at 1505:48 on low- low steam generator level just before the reactor was topped manually. The turbine driven auxiliary feedwater pump automatically started on low-low steam generator levels. The Train A and Train B motor driven auxiliary feedwater pumps had already been manually started. At approximately 1506 a feedwater isolation signal occurred due to low average reactor coolant system temperature after reactor trip. However, as noted above, the feedwater isolation valves were already closed.

The control room operators dealt effectively with this event. During the event the operators considered transferring the feedwater pumps to manual General Electric speed control; however, the potentiometers were not balanced with the Westinghouse speed controller output. Additionally, the operators only had 18 seconds between the time the feedwater pumps started coasting down and the time the feedwater isolation valves closed. Overall, there was little chance of recovering from the transient. After the reactor trip, the operators restored steam generator levels at a controlled rate, ensuring reactor coolant system temperature was maintained. All systems operated as designed, allowing plant stabilization in Mode 3, Hot Standby, without further event.

An event or condition that results in an automatic actuation of any ESF, including the RPS is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At approximately 1633 on May 9, 1990, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

Not applicable - no structures, systems, or components were inoperable at the stan of the event that have been determined to have contributed to the event.

D. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE IF KNOWN

Not applicable - no component or system failures have been identified.

E. FAILURE MODE, MECHANISM,AND EFFECT OF EACH FAILED COMPONENT

Not applicable - no component failures have been identified.

F. FOR FAILURES OF COMPONENTS WITH MULTIPLE FUNCTIONS, LIST OF SYSTEMS OR SECONDARY FUNCTIONS THAI WERE ALSO AFFECTED

Not applicable - no failures of components with multiple functions have been identified.

G. FOR FAILURES THAT RENDERED A TRAIN OF A SAFETY SYSTEM INOPERABLE. AN ESTIMATE OF THE ELAPSED TIME FROM THE DISCOVERY OF INOPERABILITY UNTIL THE TRAIN WAS RETURNED TO SERVICE

Not applicable - no failures were involved.

H. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR
PROCEDURAL ERROR

During the post trip evaluation it was determined that the I&C calibration procedure was inappropriate for Mode 1.

I. CAUSE OF THE EVENT

The reactor trip occurred as a result of jumpering out the Westinghouse feedwater pump speed controllers while they were being used to maintain feedwater pump speed.

Root Cause 1

The inadequate review and approval process for certain procedure changes resulted primarily from the practice of allowing procedures to be performed in plant modes other than those specified in the prerequisite step and marking it "N/A" (Not Applicable). This practice is allowed by the I&C Work Control procedure which requires a review of technical specifications and commitments, and with Shift Supervisor concurrence to perform a procedure in a mode other than that specified in the prerequisites. Neither the I&C engineer nor the Unit Supervisor performed a complete technical review since it was not required. While the requirements of the I&C Work Control procedure were fully satisfied, a more detailed review would have identified the need for some other method of calibration in Mode 1.

Root Cause 2

The plant impact review of the work order by Operations and I&C was less than adequate. The procedure's special conditions state the Feedwater Pump A speed controller, the Feedwater Pump B speed controller, and associated alarms should be considered out of service during the test. This should have alerted

Operations and I&C to the potential for a loss of feedwater pump speed control. The Westinghouse speed controllers were still being used when the event was initiated.

Contributing Factor 1

The I&C calibration procedure was less than adequate because it was written for use during cold shutdown, and did not clearly identity the plant impact in the prerequisites section. The procedure states that during this test, the Feedwater Pump A speed controller and the Feedwater Pump B speed controller would be out of service. More descriptively, the procedure should have listed specific tag numbers stating that the Westinghouse manual and automatic speed controllers would be out
of service. Additionally, the jumpers do not need to be installed during a transmitter calibration.

Contributing Factor 2

Prior to installation of the jumpers, the I&C technician did not review applicable documentation to understand the impact of the installed jumpers. Therefore, when the I&C technician installed the jumpers, he did not know the effect of the jumpers nor was he aware of their purpose.

J. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety systems actuation signals occurred as a result of the event.

Feedwater Isolation (EIIS:(SJ))

Reactor Protection System (EIIS:(JC))

Auxiliary Feedwater (EIIS:(BA))

K. FAILED COMPONENT INFORMATION

Not applicable - no failed components were involved.

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS
EVENT.

The loss of speed control and coastdown of both feedwater pumps is a loss of normal feedwater event. The Loss of Normal Feedwater analysis in the Final Safety Analysis Report (FSAR) Chapter 15.
.7
would bound the event for several reasons. (1) The FSAR analysis is performed with assumptions which minimize the decay heat removal capability (i.e., secondary system steam relief through the self-actuated safety valves instead of the steam dumps, and the worst single failure in the auxiliary feedwater system). In the event, the motor driven auxiliary feedwater pumps were started before steam generator water levels reached the low-low thp point. The Auxiliary Feedwater System performed without failure. (2) The FSAR analysis assumes the initial reactor coolant average temperature is 6.5 degrees F higher than the nominal Engineered Safety Features (ESF) value. The plant was at normal average temperature at the time of the event. (3) The FSAR analysis is performed at 102 percent of thermal rated power The plant was operating at 48 percent power at the time of the event. These assumptions conservatively bound the event at all expected power levels. Because this event is bounded by the FSAR Analysis and because the FSAR analysis shows that a loss of normal feedwater does not adversely affect the core, the reactor coolant system (EIIS:(AB)), or the steam system, this event posed no threat to the health and safety of the public.

Ill. CORRECTIVE ACTIONS

A. IMMEDIATE

A post trip review was performed to verify proper response of automatic protection systems, assess plant conditions and document information related to the event.

The I&C procedure for performing the transmitter calibration was changed on May 9, 1990. Steps were included which allow transmitter calibration at power, without jumpering out the Westinghouse speed controllers. The transmitter was successfully recalibrated on May 10, 1990 under a corrective maintenance work order.

B. ACTIONS TO PREVENT RECURRENCE

Root Cause 1

The review and approval process for certain procedure changes was inadequate.

Root Cause 1 Corrective Action

The I&C Work Control procedure will be revised to ensure that will be performed whenever a mode applicability interpretation is done until the procedure revision is implemented; I&C has issued a memorandum requiring mode prerequisite changes to be processed as procedure changes, in lieu of marking prerequisite "N/A".

Root Cause 2

The lack of a more thorough review of maintenance impact on plant operations.

Root Cause 2 Corrective Action

A Lessons Learned Information Form was issued detailing specific questions the operators should have satisfactorily answered with respect to the impact on the plant, prior to allowing work to be performed.

The operators were reminded via shift order that they may send a priority 12 work package to Site Work Control Center to conduct a plant impact assessment similar to what is currently being done with routine work by the Site Work Control Center.

I&C will conduct training on the Lessons Learned Information Form to ensure that the question list is understood and addressed prior to the Shift Supervisor authorizing the work.

Contributing Factor 1

The lack of a calibration procedure written for use during operating modes.

Contributing Factor 1 Corrective Action

To ensure the need for future mode prerequisite reviews is minimized, the mode applicability of each I&C procedure will be evaluated during the normal biannual review of procedures. If a procedure can be performed during any additional modes, the procedure will be revised to establish appropriate conditions for each mode added.

Contributing Factor 2

A less than adequate understanding of procedure impact prior to performance.

Contributing Factor 2 Corrective Action

I&C will conduct training that will reinforce the practice of reviewing the effects of installing jumpers, lifting leads or temporarily changing plant configuration prior to performing a maintenance or surveillance activity.

C. ACTION TAKEN ON GENERIC CONCERNS IDENTIFIED AS A DIRECT RESULT OF THE EVENT

Generic Considerations

1) Other departments in the Operations organization may not be consistent in their guidelines for "N/A of procedure steps".

2) A lack of sensitivity to the impact of non-safety components to an operating plant was identified to have existed before this event. Plant Evaluation had performed a Single Point Failure Analyses on systems with a high potential for causing a reactor trip. These analyses identified components which, by themselves, could result in a reactor trip. These analyses were not being effectively utilized.

Corrective Actions

1) To ensure a consistency within Operations,all Operation Managers will review their procedures governing "N/A of procedure steps" to ensure that appropriate checks are made prior to marking a prerequisite step N/A. These checks will include as a minimum a check of technical specifications impact on the plant and a supervisors concurrence prior to acceptance of the N/A'd step. Instructional steps to be marked not applicable will

require as a minimum the supervisors concurrence. Additionally, the N/A practice will be documented and justified on the procedure when performed.

2) The Single Point Failure Analyses will be reviewed by Technical Support, to add the necessary warnings within the Managed Maintenance Computer Program. Operations and the work groups will review their respective programs to enhance sensitivity to these components. Each manager will conduct appropriate training to ensure work on these components is more rigorously controlled and appropriate precautions are taken before performing any work on these components.

IV. PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported pursuant to 10CFR50.73.

V. OTHER

All times are approximate and Central Daylight Savings Time (CDT).

ATTACHMENT 1 TO 9006140199 PAGE 1 OF 1

Log # TXX-90189
File # 10200
907.3
Ref. # 50.73
50.73(a)(2)(iv)

June 8, 1990

W. J. Cahill
Executive Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
REACTOR PROTECTION SYSTEM ACTUATION
LICENSEE EVENT REPORT 90-013-00

Gentlemen:

Enclosed is Licensee Event Report 90-013-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Due to Loss of Feedwater Pump Speed Controllers During Maintenance." Sincerely,

William J. Cahill, Jr.

DEN/daj

Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

400 North Olive Street LB 81 Dallas, Texas 75201

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